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Waterford 3

10 CFR 50.73

W3F1-2013-0011

March 21, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Licensee Event Report (LER) 2013-001-00
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Entergy is hereby submitting Licensee Event Report (LER) 2013-001-00 for Waterford Steam Electric Station, Unit 3 (Waterford 3). This report provides details associated with an automatic reactor trip due to closure of a feedwater regulating valve.

Based on plant evaluation, it was determined that this condition is reportable pursuant to 10 CFR 50.73(a)(2)(iv)(A).

This report contains no new commitments. Please contact Chester Fugate, Licensing Manager, at (504) 739-6685 if you have questions regarding this information.

Sincerely,

Chester Fugate

CF/WH

Attachment: Licensee Event Report 2013-001-00

cc: Mr. Elmo E. Collins, Jr., Regional Administrator
U.S. NRC, Region IV
RidsRgn4MailCenter@nrc.gov

U.S. NRC Project Manager for Waterford 3
Kaly.Kalyanam@nrc.gov

U.S. NRC Senior Resident Inspector for Waterford 3
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**Attachment to**

**W3F1-2013-0011**

**Licensee Event Report 2013-001-00**

| NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 EXPIRES 10/31/2013 |
|---|---|
| **LICENSEE EVENT REPORT (LER)**<br>(See reverse for required number of digits/characters for each block) | Estimated burden per response to comply with this mandatory information collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection. |

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Waterford 3 Steam Electric Station | 05000 382 | 1 OF 6 |

**4. TITLE**
Main Feedwater Regulating Valve Closure Results in Automatic Reactor Trip

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER 05000 |
| 01 | 21 | 2013 | 2013 | - 001 - | 00 | 03 | 22 | 2013 | FACILITY NAME | DOCKET NUMBER 05000 |

| 9. OPERATING MODE | 11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) | | | |
|---|---|---|---|---|
| 1 | ☐ 20.2201(b) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(i)(C) | ☐ 50.73(a)(2)(vii) |
| | ☐ 20.2201(d) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(ii)(A) | ☐ 50.73(a)(2)(viii)(A) |
| | ☐ 20.2203(a)(1) | ☐ 20.2203(a)(4) | ☐ 50.73(a)(2)(ii)(B) | ☐ 50.73(a)(2)(viii)(B) |
| | ☐ 20.2203(a)(2)(i) | ☐ 50.36(c)(1)(i)(A) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(ix)(A) |
| 10. POWER LEVEL | ☐ 20.2203(a)(2)(ii) | ☐ 50.36(c)(1)(ii)(A) | ☒ 50.73(a)(2)(iv)(A) | ☐ 50.73(a)(2)(x) |
| 91 | ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(v)(A) | ☐ 73.71(a)(4) |
| | ☐ 20.2203(a)(2)(iv) | ☐ 50.46(a)(3)(ii) | ☐ 50.73(a)(2)(v)(B) | ☐ 73.71(a)(5) |
| | ☐ 20.2203(a)(2)(v) | ☐ 50.73(a)(2)(i)(A) | ☐ 50.73(a)(2)(v)(C) | ☐ OTHER Specify in Abstract below or in NRC Form 366A |
| | ☐ 20.2203(a)(2)(vi) | ☐ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (*Include Area Code*) |
|---|---|
| Waterford 3 Steam Electric Station Chester Fugate | (504) 739-6685 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| X | LD | NA | NA | N | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|
| ☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE) ☒ NO | | | | |


**ABSTRACT** (*Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines*)

On January 21, 2013 at 15:51 CST, Waterford 3 experienced an automatic reactor trip from approximately 91% power due to lowering Steam Generator (SG) #1 level following the unexpected closure of Main Feedwater Regulating Valve #1 due to an instrument air line failure. Emergency Feedwater (EFW) Actuation Signals (EFAS-1 and EFAS-2) were received due to low SG levels, which is an anticipated response to the reactor trip with the plant at or near full power. SG #1 received EFW system flow for a short period of time. The plant stabilized in Mode 3 with levels in both SG's restored to normal operating band with the Main Feedwater (MFW) system. Adequate water level was maintained in the SG's during the transient to ensure decay heat removal from the Reactor Coolant System (RCS). This condition did not compromise the health and safety of the general public.

This condition is reportable pursuant to 10CFR50.73(a)(2)(iv)(A) due to the automatic actuation of the Reactor Protection System (RPS) and due to the automatic actuation of the EFW system.

NARRATIVE

REPORTABLE OCCURRENCE
On January 21, 2013 at 15:51 CST, Waterford Steam Electric Station, Unit 3 (Waterford 3) experienced an automatic reactor trip from approximately 91% power due to lowering Steam Generator (SG) [SG] #1 level following the unexpected closure of Main Feedwater Regulating Valve (MFRV) [FCV] #1. Emergency Feedwater (EFW) [BA] Actuation Signals (EFAS-1 and EFAS-2) were received due to low SG levels which is an anticipated response to the reactor trip with the plant at or near full power. The condition was reported to the NRC Operations Center within four hours under criteria 10CFR50.72(b)(2)(iv)(B) for an automatic reactor trip of the plant while the reactor was critical. Additionally, the condition was reported under criteria 10CFR50.72(b)(3)(iv) for the automatic actuation of EFAS upon low SG levels (an eight hour reporting requirement). This condition is reportable as a Licensee Event Report pursuant to 10CFR50.73(a)(2)(iv)(A) due to the automatic actuation of the Reactor Protection System (RPS) and due to the automatic actuation of the EFW system.

BACKGROUND
Waterford 3 is a Combustion Engineering design pressurized water reactor with two SG's. Both of the recirculating type U tube SG's and the reactor vessel closure head [RPV] were replaced during the recently completed refueling outage RF-18.

INITIAL CONDITIONS
Waterford 3 had recently completed RF-18 and was in the process of raising plant power to 100%. Plant operation was being conducted using normal plant operating procedures. There were no Technical Specification Limiting Conditions of Operation specific to this condition in effect. During the power escalation, plant personnel had noted that MFW system vibrations were higher than had been previously experienced and had put a plan in place to measure and evaluate the condition.

EVENT DESCRIPTION
On January 21, 2013, during power escalation following RF-18, plant power was stabilized at approximately 91% reactor power to facilitate placing the three non-safety heater drain pumps in service. Heater Drain Pump [P] C was started and had been running approximately five minutes when the heater drain pump tripped. Concurrently, Control Room operators reported that SG#1 level was dropping. The operating crew entered the Steam Generator Level Control Malfunction procedure. Per the procedure, the Control Room Supervisor directed taking manual control of SG #1 level and restoring SG level to normal operating band. Operators communicated that MFRV #1 indicated near full closed and they were unable to operate the valve. Before further action could be taken, an automatic reactor trip occurred due to SG #1 level lowering to setpoint. EFW Actuation Signals (EFAS-1 and EFAS-2) were received due to low SG levels, which is an expected response to the reactor trip with the plant at or near full power.

The operating crew entered and performed the Standard Post Trip Actions procedure. The safety related EFW system pumps started in response to the EFW actuation signals. SG #1 received EFW system flow for a short period of time. The plant stabilized in Mode 3 at normal operating temperature and pressure. Levels in both SG's were promptly restored to normal operating band with the non-safety Main Feedwater (MFW) [SJ] system. MFRV #1 was noted to be closed, which is expected for normal post trip response. The operating crew transitioned to the Reactor Trip Recovery procedure and verified that all safety function criteria were met. The EFW actuation signals were reset.

Following the event, a post trip review was performed in accordance with plant procedures. The review determined that the plant had responded as expected to the transient and that the Reactor Protection System (RPS) [JC] and Engineered Safety Features Actuation System (ESFAS) [JC] had performed as designed. The condition was entered into the site corrective action program as CR-WF3-2013-0445.

CAUSAL FACTORS

During the startup from the RF18, MFW line vibration levels were much higher than expected. A vibration plan had been developed for use during plant startup. The plan was based on a similar plan developed for a previous power uprate. The MFW line vibration levels were acceptable following the power uprate and little increase was expected from replacement of the steam generators.

A root cause evaluation was performed to determine the cause of MFRV #1 failing closed. The evaluation concluded that the high MFW line vibration levels experienced after steam generator replacement caused the cantilevered instrument air [LD] line supplying MFRV #1 to fail resulting in valve closure which lowered MFW flow to SG #1 and caused a subsequent automatic reactor trip on low steam generator level.

The evaluation reviewed plant changes performed in RF18 to determine if any plant modifications had the potential to impact MFW line vibration. The only modification identified was the steam generator replacement. A detailed change analysis was performed to determine the major changes between the old and new steam generator design. Although several differences were evaluated, this analysis was only able to determine possible causes of the increases MFW line vibration.

- The most likely possible cause of increased feed water vibration is a change in the acoustic characteristics of the combined feedwater / steam generator system changing as a result of the installation of the replacement steam generators (RSG).

    The most likely possible cause of increased feed water vibration is that the acoustic response of the RSG is leading to an induced pressure pulse which is harmonic with the piping's natural frequency which is causing elevated system response (i.e., vibration) to dissipate the energy. The RSG may be either the cause of this phenomenon, i.e. fluid conditions inside the steam generator may be causing pressure pulsations, or the RSG may be reflecting existing pressure pulses back into the feedwater system resulting in increased vibrations. The pressure pulsations may be influenced by higher than expected SG differential pressure. Pulsation driven vibrations directly contributed to the initial failure of the MFRV #1 air line and subsequent reactor trip.

- A possible cause of increased feedwater system vibration is the combination of MFRV position and increased feedwater pump speed.

    A second possible cause of increased feedwater system vibration is the combination of MFRV position and RSG increased pressure causing increased feedwater pump speed resulting in increased system head and subsequent head loss between the MFRV and RSG. The resulting energy loss is manifested as increased vibration.

- A contributing cause is that MFRV valve operator design is susceptible to vibration induced failure.

  MFRV #1 and #2 were susceptible to vibration induced failure. The configuration of both valves utilized an air booster relay mounted to a hard pipe a distance off the actuator housing. This configuration was susceptible to the changed vibration conditions experienced during the plant's startup from RF18.

- A contributing cause is that the initial pre-planned vibration monitoring and mitigation plan was not adequate to detect and mitigate unanticipated problems.

  The initial pre-planned vibration monitoring plan was not adequate to detect unanticipated problems. Previous steam generator replacement vibration issues were known by the steam generator replacement project personnel and had been considered in developing a vibration monitoring plan. However, sufficient rigor was not applied in the use of Operating Experience. The risk at Waterford was evaluated. Since the risk of a substantial rise in vibration was considered low, actions were not put in place to perform detailed walkdowns of the Waterford 3 systems for vulnerabilities to the potential for increased vibration and hardening of the system at those points. At power levels below 80% the monitoring plan relied upon installed plant instrumentation. At 100% power the collection of data at pre planned monitoring points (route monitoring) was to begin. General field walkdowns to look for unanticipated problems were not performed. Following the plant trip the vibration plan was expanded to include more route points as well as taking data at more power plateaus (35-45%, 50%, 80% and 100%) and to include routine operator walkdowns.

EXTENT OF CONDITION

Following the reactor trip due to low Steam Generator #1 level on 1/21/2013, it was determined that MFRV #1 had failed closed due to an instrument air line failure. This condition did not allow air to be supplied to the actuator to open or throttle the valve as required to restore or maintain level as demanded by the Main Feed Water Control System. The instrument air line was examined and it was determined the failure was due to fatigue at the threaded connection. There was evidence some damage was already present and it was surmised a possible rise in vibration due to system dynamics changing from the replacement of steam generators may have aggravated the condition.

Following the plant trip, the air line material was examined by engineering and the material used was schedule 40 piping. The air line was repaired with the original schedule piping. The unit returned to power operations and, at 100% reactor power, engineering had a vibration technologist performing vibration surveys. While taking vibration data on MFRV #1, the technician discovered that the new instrument air line had begun failing and notified Operations that immediate assistance was required before complete failure occurred. The failure of the air line was in the same location where the original air line had failed. The air line was once again replaced and vibration values were obtained which were greater than anticipated.

The same instrument air line on MFRV #2 was identified as having schedule 40 piping and was proactively replaced; both instrument air lines were subsequently reconfigured with flexible tubing.

NARRATIVE

CORRECTIVE ACTIONS

- Implement revised vibration monitoring plan (complete)
- Perform a walkdown of feedwater lines to identify any additional vibration concerns (complete)
- Perform a walkdown of feedwater lines to identify any additional concerns with supports (complete)
- Develop and implement engineering changes to mitigate vibration effects on selected components (complete)
- Develop additional guidance to Operations Department as an Operational Decision Making Issue (ODMI)(complete)
- Obtain the pressure profile including pressure pulses in the FW system and vary the feed pump speed at constant feedwater flow to determine the effect on vibration (in progress)
- Have selected expert review the result of the pressure testing profiles. Utilize pressure pulsation frequency and magnitude to evaluate potential effects on RSG internals including tube bundle and feed ring (planned)
- Perform a review of the pressure testing profiles. Determine if the plant's response is within expected bounds for plant systems (planned)

SAFETY SIGNIFICANCE

During Cycle 19 plant start-up and power escalation at approximately 90% Rated Thermal Power (RTP), the plant experienced a reactor trip due to the closure of MFRV #1 and the subsequent drop of narrow range SG-1 water level. The failure of MFRV #1 was due to an instrument air line break on the valve's actuator. This event caused a loss of normal feedwater flow to SG-1 which is bounded by the existing FSAR licensing basis analysis. FSAR 15.2.2.5 discusses the total loss of a normal feedwater flow event. A loss of normal feedwater flow is defined as a reduction in feedwater flow to the steam generators when operating at power, without a corresponding reduction in steam flow from the steam generators. This flow imbalance results in a reduction in the steam generator water inventory and a consequent heat up of the reactor coolant. The complete loss of normal feedwater flow is analyzed by assuming an instantaneous stoppage of feedwater flow to both steam generators. The complete loss of normal feedwater case is analyzed since this condition requires the most rapid response from the Plant Protection System (PPS) [JC]. The analysis results demonstrate that the total loss of feedwater flow does not challenge the DNBR and fuel temperature SAFDLs; the DNBR and LHR limits are not exceeded. Thus, the failure and closure of the main feedwater regulating valve event is bounded by the FSAR loss of normal feedwater analysis.

In addition, the FSAR 15.2.3.1 discusses the feedwater line break event analysis. FSAR 15.2.3.1 shows that acceptable results are achieved for small and large break sizes in the feedwater piping. A feedwater line break is also bounded for its impact on containment pressure by the main steam line break event, as stated in FSAR 6.2.1.1.3.

This event did not result in any nuclear safety consequences.

This event did not result in release of any radioactive material and therefore had no radiological safety consequences.

SIMILAR EVENTS

A search was performed for other similar reported events at Waterford 3. No similar events were identified.

NARRATIVE

ADDITIONAL INFORMATION
Energy industry identification system (EIIS) codes are identified in the text with brackets [ ].